Third Meeting of the ITPA Coordinating Committee, 24-25 October 2002, IPP-Garching, Germany

RF Tokamak Program Status and Plans Report

Presented by N. Ivanov RRC Kurchatov Institute

T-10 Experiments

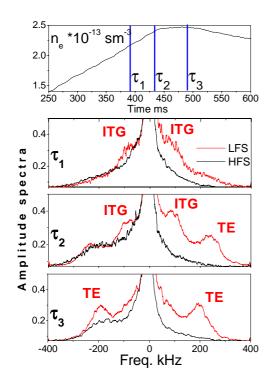
(Kurchatov Institute)

Turbulence Studies

The studies of turbulent spectra in the ECR heated discharges at EC power up to 1.4 MW using the heterodyne O-mode correlation reflectometer and multipin Langmuir probe.

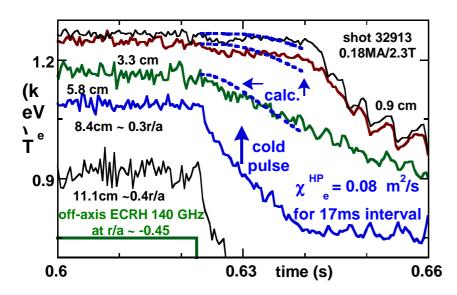
The change of turbulence characteristics from the Core Type to the Edge Type was found at a certain radius that increases with plasma current.

- The Core Type Turbulence is attributed to the Ion Temperature Gradient and Trapped Electrons Instabilities.
- The Edge Type is attributed to Resistive Ballooning Interchange Mode.



Fourier spectra of the Core Turbulence for flat (τ_I) and peaked (τ_3) plasma density profiles.

Study of the Electron Heat Pulse Propagation (HPP) with 130+140 GHz Gyrotrons in sawteeth-free low-shear plasmas formed by off-axis ECRH



Inward cold HPP created by switching-off of the off-axis ECRH:

 $\chi_e^{HP} \sim 0.1 \text{ m}^2/\text{s inside}$ r/a < 0.3,

 $R \nabla T_e / T_e$ is up to 17.

Outward HPP from 0.5 MW on-axis ECRH switching-on:

 $\chi_e^{HP} \sim 0.2\text{-}0.3 \text{ m}^2/\text{s} \text{ at } 0.2 < r/a < 0.37,$

 $R \nabla T_e / T_e$ is up to 23,

Ohmic level: $\chi_{eOH}^{HP} \sim 0.6 \text{ m}^2/\text{s}$.

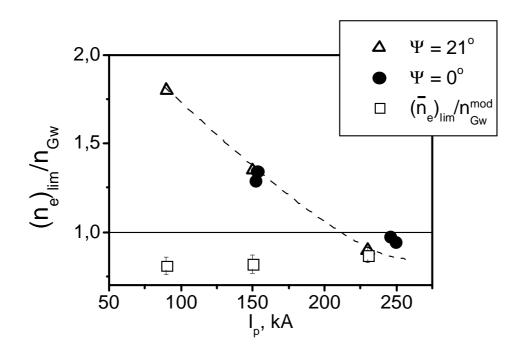
Off-axis-ECRH improved confinement zone: in low-shear region near q=1.

Plans

To clarify q=1 role by variation of scenario and increasing the ECRH power.

High Density Experiments with Strong Gas Puffing under ECRH

In T-10 conditions the Greenwald limit can be exceeded by the factor 1-1.8 depending on the plasma current.



$$n_{Gw} = I_p / (\pi a^2)$$

$$n_{Gw}^{mod} = I_p / (\pi a_{cur}^2),$$
where $a_{cur} = r$ (I=0.95I_p)

- Oblique power launch Ψ=21°
- Perpendicular power launch Ψ=0°

Plasma Biasing

Study the radial electric field effect on the plasma turbulence and confinement.

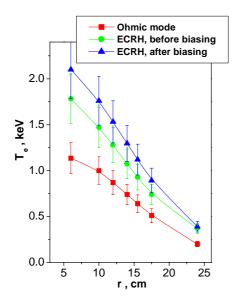
Positive voltage (+450 V) was applied to the electrode inserted into the plasma (2 cm).

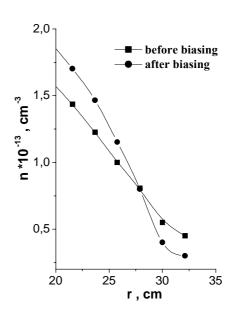
In ECRH regimes, the L-H transition is observed at the ECRH power below the threshold of the spontaneous L-H transition. The transition is accompanied by increases of

Electron temperature (15%),

Ion temperature (30%),

Plasma density (30%).

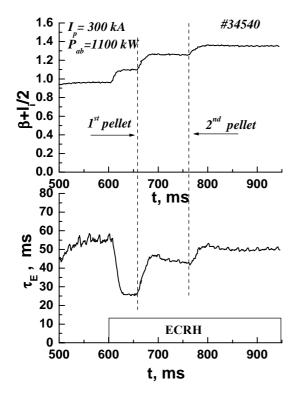




Electron temperature and plasma density profiles before and after biasing.

Plasma Confinement under EC Heating and Pellet Injection

The behaviour of plasma after deuterium pellet injection is similar to L-H transition. The decrease of plasma turbulence and formation of high density gradient zone are observed.



Improvement of the plasma confinement under consequent pellet injections

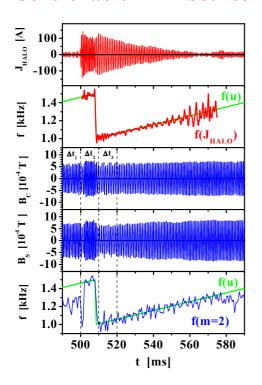
Plans

The experiments will be continued with higher EC heating power (2.5 MW) and a multiple (up to 8) pellet injection.

Investigation of the Controlled-Halo-Current Effect on the Plasma MHD-Activity

Halo-current loop:

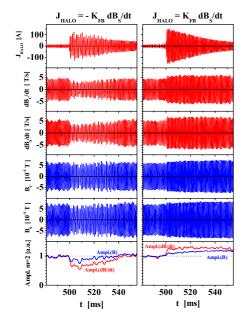
Rail Limiter - Plasma - Vacuum Vessel - External Circuit with a Controllable EMF Source - Rail Limiter.



Preprogrammed Control Signal

The m=2 mode frequency follows the halo-current frequency variations.

The frequency of the control signal, f(u), is shown together with the halo-current frequency, $f(J_{HALO})$, and the m=2 mode frequency, f(m=2).



Feedback Control

The effect of the halo-current on the mode behaviour depends on the choice of the feedback phase shift.

Plans Experiments with improved feedback circuit.

Design of T-15M Tokamak

(Efremov Institute, Kurchatov Institute)

2002

- Technical Design of Main Systems is completed
- Project works (cooling systems, communications, automatics and etc.) have started

2003

• Final Design of Main Systems

2004-2008

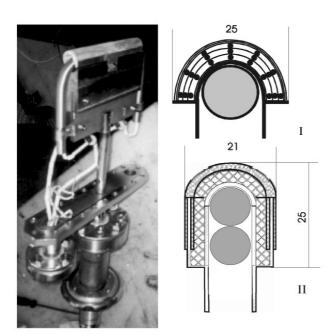
• T-15M Construction



Plasma current, I _P , MA	1.7
Aspect ratio, A	3.1
Major radius, Ro, m	1.55
Minor radius, a, m	0.5
Elongation, k ₉₅	1.7
Triangalurity, δ ₉₅	0.35
Toroidal field, B _T , T	2.5
Plasma current duration, ∆t, s	5
Auxiliary heating power, Paux, MW	20
Auxiliary heating duration, s	2
Plasma shape	SN

T-11M Experiments (TRINITI)

Plasma Interaction with Lithium Limiter



Experiments with 0.1 sec tokamak discharges.

- ➤ Lithium CPS material of the limiter does not fail and does not loose a noticeable amount of lithium at thermal load of 10MW/m².
- ➤ Under T-11M conditions, the lithium losses are caused by:
- Ion sputtering at T_s<500 C,
- Evaporation at T_s>500⁰C.

General view of the First Lithium rail T-11M limiter. Cross-sections of first and second variants of the limiter.

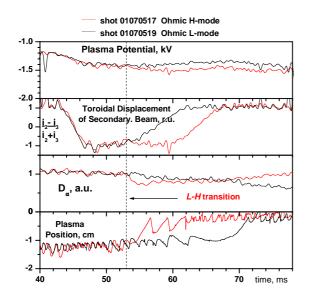
Plans

Experiments with 0.3 sec tokamak discharges and transition to stationary cooling Li-limiter regime

TUMAN-3M Experiments

(Ioffe Institute)

L-H and H-L Transitions



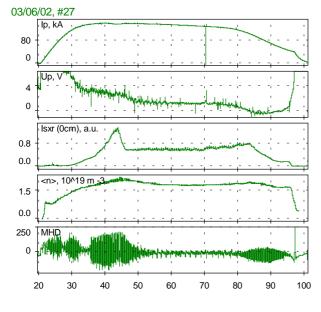
Measurement of the plasma potential evolution through the L-H transition using Heavy Ion Beam Probe

- Core plasma potential reduces by 150 V after the L-H transition with time scale of 10 ms, which is much longer than the transition time,
- The observed potential behaviour is attributed to a double-layer formation. Edge turbulence evolution is in agreement with this hypothesis.

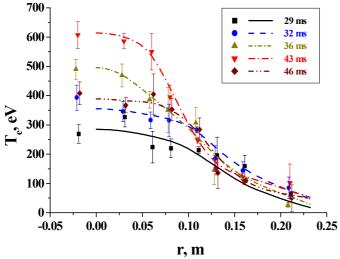
Poloidal magnetic flux perturbation effect on L-H and H-L transitions

- Positive/negative magnetic flux perturbation leads to L-H/H-L transition,
- Flux perturbation results in radial current excitation that helps in radial electric field E_r emerging. Negative E_r increases E_r shear, thus helps in the H-mode establishing. Positive E_r helps H-mode termination.

Electron Transport Barrier Formation in the Initial Phase of the Ohmic Discharge



Increase in the SXR intensity is an indication of the transport barrier formation



The $T_e(0)$ maximum during the improved confinement phase is 50% higher than $T_e(0)$ in the later phase of the discharge

Plans

- Investigation of the confinement inside the magnetic islands,
- Study of e-ITB formation conditions in ohmically heated plasma,
- Transport-barrier formation by deuterium pellet injection,
- Measurement of radial electric field and plasma rotation at L-H transitions.

FT-2 Experiments

(Ioffe Institute)



R= 55 cm, a = 8 cm, $B_T = 2.2$ T, $I_p = 20-30$ kA, $t_{plasma} = 60$ ms.

Lower-Hybrid system:

Two-waveguide Grill $N_{\parallel} = 2.5 - 3$,

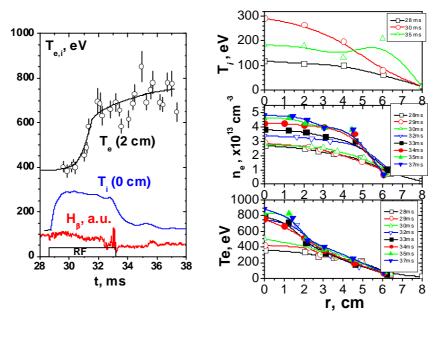
 $P_{RF} = 200 \text{ kW},$

 $f_{RF} = 920 \text{ MHz},$

 $t_{RF pulse}$ = 3-10 ms.

- > LH current drive and LH heating,
- ➤ Improved confinement and transport barriers generated by ion heating,
- ➤ Microwave diagnostics of small-scale turbulence,
- > Transport in the Scrape of Layer (SOL) and near Last Close Flax Surface (LCFS).

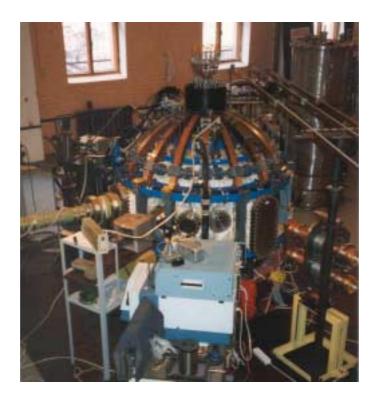
Lower Hybrid plasma heating related both to the direct RF power absorption and plasma transport suppression.



Plans

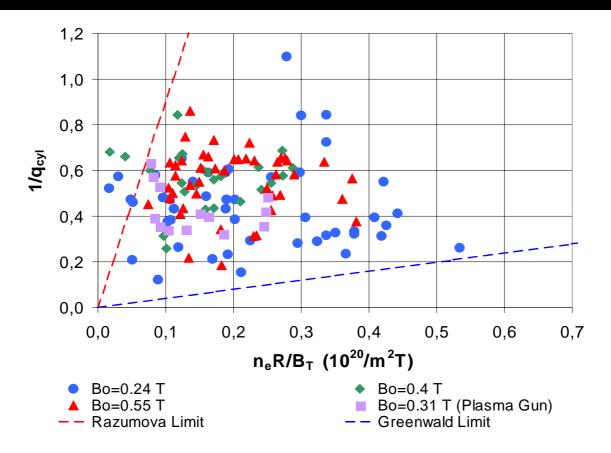
Experiments on Lower-Hybrid current drive, plasma - wave interaction and transport-barrier formation under LH Heating will be continued.

GLOBUS-M Experiments (Ioffe Institute)

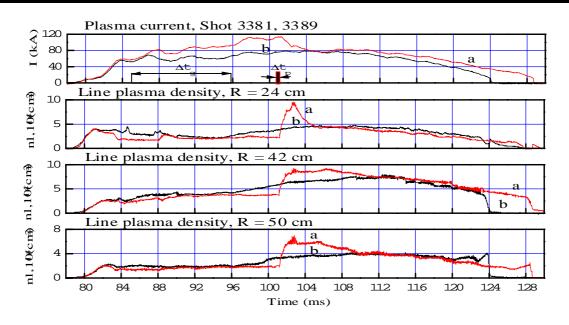


		Design
		values
Toroidal magnetic field [T]	0.54	0.62
Plasma current [MA]	0.35	0.5
Major radius [m]	0.36	0.36
Minor radius [m]	0.24	0.24
Elongation	1.9	2.2
Triangularity	0.45	0.3
Line-average plasma density [m ⁻³]	$\leq 0.4 \cdot 10^{20}$	1.10^{20}
Pulse length [s]	0.08	0.2

Stability and operational limits in OH regime



Plasma Gun Fuelling



Fast density increase (~0.5 ms) accompanied the injection of plasma clusters into the plasma.

Parameters of the clasters: Plasma density: 10^{22} m⁻³,

Total number of injected particles: 10^{17} - 10^{21} m⁻³,

Plasma cluster velocity: 30 km/s.

Plans

- Increase of the FW (f = 25-30 MHz) heating power up to 0.5 MW,
- Plasma fuelling with increased to 70 100 km/s velocity of plasma clusters injected into plasma.

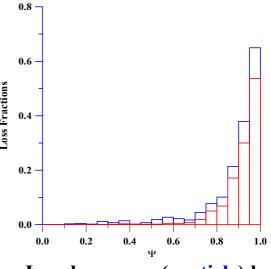
Theory and Modelling

Analysis of the combined effect of fast ion instabilities and TF ripples on fast ion loss in ITER

(Kurchatov Institute, Efremov Institute)

Ripple Losses in ITER

- Optimised Ferromagnetic Inserts provide more than one order of magnitude reduction in fast ion (fusion α -s and NBI ions) ripple losses for all reference scenarios.
- Associated heat loads on plasma facing elements in the presence of FIs are small.
- TFC + FI ripples affect fast ion transport in the very narrow layer at the outer periphery, so that possible interference of ripple assisted transport with that one due to any MHD activity is strongly suppressed.



Local α power (particle) loss fraction in SNS scenario

- Optimised FIs provide reliable tokamak operation at lower (down to ½ of nominal) toroidal field amplitude.
- <u>Plans:</u> Evaluation of fast ion loss due to additional ripples produced by the test blanket modules (TBMs) for basic ITER scenarios.

Resistive Wall Modes and their stabilisation in ITER (Kurchatov Institute)

RWM Feedback Systems with Different Input Signals

Three cases have been analysed:

- Radial field sensors,
- Poloidal field sensors inside the vessel,
- Poloidal sensors outside.

The analysis explained why the system with internal poloidal field sensors is much better than that with radial field sensors. The analysis shows that the combination of poloidal and radial sensors is the best.

Ideal and Conventional Feedback Systems for RWM Suppression

- ➤ The "ideal" helical feedback coils, looking like stellarator helical coils, would produce only helical magnetic field necessary for suppression of the RWM mode (m, n). In the case of a single conducting wall, both radial and poloidal magnetic sensors can be used for feedback stabilisation.
- ➤ The "conventional" rectangular saddle coils, considered for ITER, in addition to the harmonic (m, n) required for RWM stabilisation, produce the side-band harmonic (-m, n). As a consequence, the coils excite the stable mode (-m, n):
- In case of the poloidal sensors the (m, n) mode can be stabilised for all range of the (m, n) mode growth rates Γ_m and (-m, n) mode decay rates Λ_{-m} .
- In case of the radial sensors the (m, n) mode can be stabilised only when $\Gamma_m < \Lambda_{-m}$.

Relationship between currents in 2D and 3D models of feedback coils used for active stabilisation of RWM

➤ Magnetic field produced by 6 coils, like those used in DIII-D tokamak and considered as a candidate for ITER, can be approximated with good accuracy by a single toroidal harmonic, if the currents in these coils are properly distributed.

Analytical Study of RWM Feedback Stabilisation

- An analytical model for studying the feedback control of RWM in a tokamak with single or double conducting wall was developed. The model is based on a cylindrical approximation.
- > Outer conducting shell, in the case of ITER-like double wall vacuum vessel, insignificantly reduces the RWM instability growth rate but deteriorates the feedback stabilisation.
- ➤ Error field correction system (side saddle coils) with the nominal voltage 40 V per turn is capable of stabilising the RWM for the expected range of normalised beta

Plans

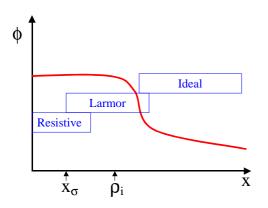
- Study of external kink mode stability limits in ITER Scenario 4 without feedback stabilisation with and without ideal 1st wall: calculation of β_N limit, effect of ports on the β_N limit (Kurchatov Institute and Keldysh Institute).
- Study of RWM feedback stabilisation in Scenario 4: calculation of RWM transfer functions (voltage applied to feedback coils => magnetic field on the probe), linear analysis of RWM feedback stabilisation (Kurchatov Institute and Keldysh Institue).
- Nonlinear analysis of RWM control on the basis of transfer functions obtained in the task A5(b), study of different feedback algorithms, analysis of current, voltage and power required for the RWM stabilisation (Efremov Institute).
- Theoretical analysis and modelling of error field effect on the RWM stability in ITER; effect of the harmonic with m/n=2/1 and amplitude of $(3-5)x10^{-5}B_T$ on efficiency of the RWM feedback control system.
- Effect of ferritic inserts, intended for reducing TF ripples, on the RWM stability and efficiency of the feedback control system will be evaluated.
- Feedback stabilisation of RWMs will be analysed taking account of dissipative plasma properties (resistivity and viscosity) and diamagnetic drift.

Analysis of NTM Onset Conditions

Generalised Transport Threshold Model of Neoclassical Tearing Modes

- Magnetic island rotation acts on the perturbed plasma temperature (density) profile inside the island in a similar way as parallel transport, i.e. it flattens perturbed plasma parameters.
- To provide weakening of the bootstrap drive of the mode the perpendicular transport should be high enough to overpower parallel transport (convection) and island rotation effect simultaneously.
- Anomalous perpendicular viscosity reduces the bootstrap drive of NTM for the islands rotating in the ion diamagnetic drift direction. For sufficiently high perpendicular viscosity, the bootstrap current effect can reverse and thus change drive for stabilisation of NTMs.
- ➤ Neither Polarisation Current nor Transport threshold models can explain wide variety of experimental data on NTM onset. New linear beta limiting "MHD sub-Larmor mode" is a candidate to cover the gap in present understanding of NTM physics.

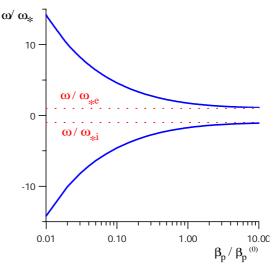
Spontaneous generation of NTMs



Spatial structure of the mode potential

For $\beta_p \approx s^2 L_p^2/r_s^2 \sim 1$, typical for all experiments showing NTMs, this mode frequency is close to characteristic diamagnetic drift frequencies. The mode can be excited by drift effects

- β threshold could originate from the development of linear non-ideal instability "MHD sub-Larmor mode"
- This possibility follows from the particular solution of the dispersion relation (two asymptotic matchings: Resistive – Larmor and Larmor – Ideal layers)



Plans

- Theoretical analysis of NTM onset conditions and possible algorithms for their stabilisation (avoidance).
- Further development of the numerical models (Kurchatov and MSU) of NTM evolution in ITER with allowing for island rotation, coupling to satellite harmonics, effects of fast particles, resistive shell and ECCD effects.

Disruption simulation in ITER

(Kurchatov Institute, TRINITI)

➤ Plasma Disruption Simulator (PDS) has been created on the basis of DINA code to simulate the full disruption scenarios in the ITER plasma. This scenarios includes the cases of Major Disruption and Hot VDE.

Plans

The model of expansion of halo area after thermal quench during disruption has to be considered with DINA fitting mode simulations of JT-60U disruption shots.

Incorporation of the SPIDER Equilibrium Code into the ASTRA Transport Code with Possibility of Ideal Stability Analysis

(Keldysh Institute of Applied Mathematics)

Self-Consistent Analysis of Plasma Transport and Ideal Stability in ITER

2D arbitrary shape equilibrium code (SPIDER) was incorporated into ASTRA code. It provides a possibility of the ideal stability analysis using KINX code in realistic plasma geometry with separatrix. The ideal ballooning and ideal external-kink-mode stability analysis has been carried out for the ITER reference steady-state scenarios. The operational limits implied by the kink mode stability on the fusion gain factor Q has been obtained for the steady state scenario.

Plans

To improve the consistency of the ideal mode stability analysis, all calculations should be done on the basis of a single transport code. Free boundary equilibrium solvers (PET and DINA codes) should be incorporated as modules in the Automatic System for Transport Analysis (ASTRA) used for ITER transport analysis.

3D Modelling of ICRF Heating and Current Drive in ITER

(Kurchatov Institute)

Hydrogen Minority ICRF Heating and Highly Localised Off-Axis Mode Converted Current Drive

ICRF scenarios are investigated for ion heating, for NTM stabilisation, for highly localised power deposition to electrons in pedestal plasma region for stabilising/destabilising ELMs through co/cntr current drive.

High Frequency Fast Wave Current Drive

Modelling of FW (200-300 MHz, $\omega >> \omega_{ci}$) current-drive in core plasma with 3D STELION code. A compact multi-waveguide antenna is used in this scenario.

Analysis of Travelling Fast Wave ICRF Antenna Radiating from a Recess in the First Wall

The concept of a travelling-wave antenna for heating and current drive in the ion cyclotron range of frequencies has been proposed and the theory of this new antenna has been developed. This concept utilises a toroidal array of short circuited poloidal loops supported by the ridge wave-guide.

The ridge wave-guide provides the required antenna dispersion properties and supports broad frequency bands to cover the operation regime of the fast wave current drive and the conventional heating scenarios.

Development of the ICRF full wave code STELION2

- > Full wave 3D ICRF STELION code was upgraded to include direct Ion Bernstein Wave (IBW) excitation at two ion hybrid resonance surfaces in ITER plasma.
- > This upgrade will give a possibility to more correctly treat RF power partition between electrons and ions and respective 3D power depositions and driven current profiles.